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TITLE: CALCULATIONS OF THERMAL-REACTOR SPENT-FUEL NUCLIDE INVENTORIES AND COMPARISONS WITH MEASUREMENTS

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Calculations of Thermal-Reactor-Spent-Fuel Nuclide Inventories and Comparisons with Measurements

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I. INTRODUCTION.

CINDER-2 is an actinide and fission-product nuclide inventory/aggregate-property summation code. The code is an outgrowth of the EPRI-CINDER code,[1] an improvement upon the original CINDER code.[2] The development of these EPRI-sponsored codes, traced in Fig. 1, has generally been directed toward the use of abbreviated libraries[3-5] in the accurate calculation of aggregate properties (e.g., fission-product neutron absorption)[6] associated with thermal reactor applications. Parallel with this development has been an evolution of the CINDER-7[7] and CINDER-10 codes and exhaustive libraries of fission-product data; these are used in calculating nuclide-inventory[8] and aggregate fission-product decay properties, including decay spectra[9] and decay heating[8] under essentially any irradiation and cooling history.

CINDER-2 development is assiciated with the development, release and utilization of the extensive ENDF/B-V actinide and fission-product data files. The formation of the supplemented ENDF/B-V library for CINDER-2 is illustrated in Fig. 2. The library includes four-group thermal-reactor-spectrum cross sections, which were collapsed with the TOAFEW-V code from the associated 154-group processed ENDF/B-V cross section library.[10] Also included are ENDF/B-V fission-product fission yield fractions, decay branching fractions, and average decay energies, adjusted where necessary to accommodate implicit nuclides and transitions in the nuclide chain structure.[11] ENDF/B-V half-lives were used to form decay constants for all common nuclides. Neutron absorption branching fractions and other data not specified in ENDF/B-V were acquired from other data sources.

The CINDER-2 ENDF/B-V based library enables the calculation of the inventory of all actinide nuclides produced in reactor fuel, as well as their aggregate decay properties.[5] The library includes data permitting the calculation of the inventory of 211 fission-product nuclides, sufficient to accurately describe aggregate neutron absorption at all times and aggregate decay properties at cooling times exceeding a few hours. A link to the EPRI-CELL code[12] permits tandem EPRI-CELL/CINDER-2 calculations of thermal reactor fuels in which CELL-calculated fluxes and density-dependent self-shielded cross sectons of the principal actinide nuclides are passed to and used in CINDER-2. These fluxes and self-shielded actinide cross sections affect the inventory calculation of all fission-product and higher actinide nuclides and are therefore required for accurate inventory calculations.

Comparisons with integral measurements have demonstrated the accuracy of CINDER codes and libraries in calculating aggregate fission-product properties, including neutron absorption,[6] decay power,[13] and decay spectra.[14,15] CINDER calculations have, alternatively, been used to supplement measured integral data describing fission-product decay power[16-19] and decay spectra.[20] Because of the incorporation of the extensive actinide library and the use of ENDF/B-V data, it is desirable to compare the inventory of individual nuclides obtained from tandem EPRI-CELL/CINDER-2 calculations with those determined in documented benchmark inventory measurements of spent reactor fuel.

II. SPENT FUEL NUCLIDE INVENTORY MEASUREMENTS AND THE ASTM METHOD E321.

Fuel isotopics measurements generally rely on one or more techniques described in the Annual Book of ASTM Standards, Part 45, for the determination of fuel burnup. The most commonly used method has been ASTM Method E321-NY, where NY is a two-digit number designating the year of original adoption (67) or revision (69, 75, 79). E321 is entitled "Standard Test Method for Atom Percent Fission in Uranium and Plutonium Fuel (Neodynmium-148 Method)." All versions of E321 define sample burnup F_T in atom percent fission using measured nuclide density ratios in expressions that reduce to the following equations:

$$F_T = F' \times 100./(U' + Pu' + F'), \tag{1}$$

$$F' = \#\ \text{fissions}/\#^{238}U\ \text{atoms}, \tag{2}$$

$$U' = \#\ U\ \text{atoms}/\#^{238}U\ \text{atoms}, \tag{3}$$

and Pu' = # Pu atoms/#^{238}U atoms. (4)

The quantity F' is determined from

$$F' = \#^{148}\text{Nd atoms}/^{148}\text{Nd yield}/\#^{238}\text{U atoms}. \quad (5)$$

All versions of E321 relate exposure (MWd/tU) and burnup (atom % fission) with

$$F_T\ (\text{MWd/tU}) = (9600 \pm 300) \times F_T\ (\text{atom \% fission}). \quad (6)$$

Revisions of E321 obviously have been made in pursuit of greater accuracy, although this may not be the result. We have not examined E321-67 or E321-69, although the latter is referred to in Ref. 21 where the ^{148}Nd cumulative yield is set at 1.68% in H. B. Robinson fuel measurements. E321-75 states that the ^{148}Nd yield should be "calculated from the fission yields of ^{148}Nd for each of the fissioning isotopes weighted according to their contribution to fission as measured in ASTM Method E244, Test for Atom Percent Fission in Uranium Fuel (Mass Spectrometric Method)." However, the paragraph continues, "For ^{235}U fuels, [the ^{148}Nd yield] can be assumed to be the fractional yield for ^{148}Nd in ^{235}U thermal fission, which is 0.01618." No appropriate yield values are given for the other fissionable nuclides. The aim here toward a "better" ^{148}Nd yield value is cancelled by the "^{235}U fuels" proviso, which is open to interpretation. Indeed, no spent fuel isotopics measurement reviewed by us to date has included a determination of fission contributions or a weighted yield fraction. All measurements have assumed the ^{235}U fission yield.

Method E321-79 lists ^{148}Nd yield fractions for all four fissionable nuclides and includes a "K" factor to adjust ^{148}Nd for nonfission production from ^{147}Nd(n,γ). The yields here are from Ref. 22, which documented the third yield set iteration en route to the fifth and final yield set used in ENDF/B-V.

The ^{148}Nd cumulative fission yield fractions of ENDF/B-IV, -V, preliminary -VI, and E321-79 are listed in Table I. Reference to "^{235}U fuels" and the defacto acceptance of the use of the ^{235}U fission yield fraction for all fissions is absent in E321-79.

The $^{147}Nd(n,\gamma)$ cross section used in calculating K is from Ref. 23, where the repo 440 b thermal cross section depends linearly on the assumed 50% intensi of the 301.7 keV neutron capture gamma ray. The argument for this 50% assumption seems weak; a model code could be used to determine a more precise intensity and thus a more precise cross section. The ENDF/B-V evaluation for ^{147}Nd lists a 2200 m/s (n,γ) cross section of 49b and resonance integral of 647.8b. The E-321-79 treatment of the K factor and the $^{147}Nd(n,\gamma)$ cross section adjusts the 440b cross section to a 300°C Maxwellian-averaged value of 247b, assuming 1/v behavior. This 1/v extension of the Ref. 23 value is compared in Fig. 3 with the ENDF/B-V representation, which was based on a model code calculation adjusted to agree with a resonance integral measurement. Regardless, no spent fuel isotopics measurement reviewed by us to date has included a determination of K or any ^{148}Nd density adjustment to correct for neutron absorption effects.

All versions of E321 assume that burnup and exposure are related by Eq. (6), which assumes that all fissions result in the realization of the same amount of heat, approximately 201.5 MeV. Our calculations of high-burnup Calvert Cliffs 1 fuel with EPRI-CELL show that the heat/fission realized, using the data of Ref. 24, increases from 201.5 MeV to 220.9 MeV at 46.8 GWd/t in 2.45% enriched fuel. This increase is due to the increase with A in recoverable energy/fission excluding capture effects, and an increase with exposure of the average decay energy produced in neutron capture by the capture products and daughters. The cumularive effect of this increase is not so drastic, but the Exposure-to-Burnup ratio still exceeds 9600 by nearly 5% at 46.8 GWd/t.

Unfortunately, complete compliance with ASTM Method E321-79 may produce different and less-accurate results than those obtained with an earlier and less intricate version. We have constructed a reduced ENDF/B-V fission product library for CINDER-2, following and recording all ^{148}Nd modes of formation and loss. We have used the library in tandem EPRI-CELL/CINDER-2 calculations of Calvert Cliffs 1 fuel to 46.8 GWd/tU. The results of the exercise are given in Table II. Note that Exposure and Burnup are listed at the left, as well as their ratio. The cumulative fission density and per cent contributions from each fissionable nuclide are then given -- these would be determined experimentally with ASTM Method E244. The ^{148}Nd formed directly by yield is then given; this is always greater than 99.1% of all ^{148}Nd produced, corresponding to K ≥ .991. The direct yield tabulated is the ratio of ^{148}Nd formed directly to the cumulative fissions; this is the desired weighted yield of E321-79. The trace ^{148}Nd formed from ^{147}Nd(n,γ) is then tabulated, as well as the gross ^{148}Nd formed by both paths.

The K factor of E321-79 is evaluated in the standard for a range of flux and fluence values, using the 274 b. cross-section value for ^{147}Nd(n,γ) and assuming continuous reactor operation. These values are given in Table III. The Calvert Cliffs 1 fuel inventory calculations described above modeled a spent fuel sample discussed in the following sections, and the power history included intermediate shutdowns and partial power operation periods. Ignoring shutdowns, the fuel sample operated at an average integral flux of ~ 2.5×10^{14} n/cm^2/s for ~ 43 000 hours and was discharged at a fluence of ~ 3.9×10^{22} n/cm^2 and an exposure of ~ 46.8 GWd/tU. This far exceeds the maximum fluence (3×10^{21}) for which K has been evaluated, and the E321-79 method provides no guidance or data for the calculation of K. At the above flux value, an interpolated value is obtained from Table III of K ≅ .910 at the maximum fluence. This corresponds to an exposure in the Calvert Cliffs 1 fuel of ~ 3.6 GWd/tU, where the value interpolated from the calculated (direct % gross) values of Table II is K ≅ .994. The K factor of E321-79 indicates that, at an exposure of 3.6 GWd/tU, 9% of the ^{148}Nd formed has been produced from the ^{147}Nd(n,γ)^{148}Nd path. CINDER-2 calculations show that, at this low exposure, only 0.6% of the ^{148}Nd formed is from this path. These different contributions reflect the different cross-section values and/or flux interpretation used in their calculation.

No mention has been made of the ^{148}Nd loss by ^{148}Nd(n,γ), listed in Table II. Note that the cumulative ^{148}Nd loss by the ^{148}Nd(n,γ) reaction exceeds the cumulative ^{148}Nd gain from ^{147}Nd(n,γ) at exposures exceeding ~ 24 GWd/tu. The net ^{148}Nd [gross - ^{148}Nd(n,γ) loss] and the net yield are the rightmost entries of Table II. Note that the calculated net yield varies only slightly during exposure, indicating that, for this fuel, increases in ^{148}Nd, due to the increase with exposure in the weighted cumulative fission yield fraction [i.e., mass-148 yield from fission] and ^{147}Nd(n,γ)^{148}Nd production (both recognized in E321-79), are offset by the ^{148}Nd(n,γ)^{149}Nd loss that is not recognized in E321-79. Of course, these observations depend upon the accuracy of ENDF/B-V cumulative fission yield fractions and evaluated cross sections of both ^{147,148}Nd.

III. SURVEY OF AVAILABLE LWR SPENT FUEL NUCLIDE INVENTORY MEASUREMENT RESULTS

A nuclide inventory measurement of benchmark quality might well include the following:

1. A full description of the fuel physical parameters (e.g., enrichment, pellet density, pellet diameter, clad thickness and material, pitch, etc.) and environment (e.g., core location, proximity to control rods, burnable poisons, etc.).
2. A value of sample burnup and/or exposure, as well as all measured nuclide ratios and the basic data and methodology used in the determination.
3. A detailed power history of the sample plus dates of shutdown and measurements.
4. Inventory values for a wide range of nuclides.
5. Evaluated uncertainty values for all measured quantities.
6. Complete and referenceable documentation.

Unfortunately, inventory measurements are of inconsistent quality, completeness, and documentation. Measurements are characteristically funded by the utilities and the results are often proprietary. There exists no organized effort for the collection, examination, evaluation, normalization, documentation, and/or distribution of spent fuel nuclide inventory benchmark data. We encourage the Electric Power Research Institute, due to its direct association with the utilities, to assume such a function.

A preliminary list of potential LWR spent fuel nuclide inventory benchmarks is given in Table IV. Much of the information in this list is taken from Ref. 25; some of the measured data corresponding to the listed samples are presently proprietary.

IV. EPRI-CELL/CINDER-2 NUCLIDE INVENTORY CALCULATIONS AND COMPARISONS WITH MEASURED INVENTORIES

A. Methodology

Nuclide inventories are determined with tandem EPRI-CELL and CINDER-2 calculations. EPRI-CELL[12] computes the space-, energy-, and burnup-dependent neutron spectrum within a cylindrical cell of an LWR fuel rod. It uses the B_1 method of GAM[26] and the integral transport method of THERMOS[27] to produce coarse-group neutron fluxes and cross sections for subsequent depletion analysis. The temporal behavior of actinide and fission-product nuclides important to fission and/or absorption are determined with a series of linearized chains consistent with the CINDER methodology.

The EPRI-CELL model representation consists of a cylindrical fuel region surrounded by a clad region, a moderator region, and an outer "extra" region. The extra region is used to represent the environment of the fuel rod. Four radial space points are generally assigned to the fuel, one to the clad, seven to the moderator, and two to the extra region.

Data required by EPRI-CELL include infinitely-dilute multigroup cross sections, composition- and energy-dependent resonance self-shielding factors, and energy transfer matrices including upscattering in the thermal range. For convenience, these data are divided into three files. FASTLIB is a 62-group cross-section library covering the range 1.885 eV to 10 MeV for the modified GAM portion of the code. THRMLIB is a 35-group library covering the range 0.001012 eV to 1.855 eV for the modified THERMOS portion of the code. BURNLIB is a 4-group cross-section library spanning the energy range of the other libraries for use in the modified CINDER potion of the code.

EPRI-CELL generates a file of burnup-dependent collapsed 4-group flux values and, for each selected actinide nuclide, 4-group cross sections and densities at each space point. These and other data are read by a small utility program PHAZE, which prepares a CINDER-2 user input file for calculating the comprehensive nuclide inventory at any fuel space point or for the fuel average. Accuracy of the interfaced information depends upon the accuracy of the EPRI-CELL problem specification: power history, fuel description (pellet radius, density, pitch, isotopic composition, and temperature), clad description (material, inside radius, outside radius, and temperature), moderator description (% void if BWR, parts-per-million boron, and temperature) and core structure description (extra region compositon). Cooling intervals following shutdown must be input to PHAZE in order for the CINDER-2 input to include the decay to sample inventory measurements following irradiation.

The procedure of the tandem calculations must generally be repeated with power-history magnitude adjustments in order to have close agreement between a measured and calculated parameter, i.e., burnup (atom % fission), exposure (MWd/t), or some selected atom ratio (e.g., $^{148}Nd:^{238}U$, $^{137}Cs:^{238}U$, etc.). In view of our observations above on quoted sample exposure and burnup values, we have generally attempted to normalize calculations to measurements by comparing atom ratios.

B. Three Mile Island-2 Air Sample

The Three Mile Island-2 (TMI-2) unit experienced an accident early on March 28, 1979, resulting in the release from the fuel of a portion of the fission-product inventory. The accident accured after a short operating history described in monthly operating reports to NRC from the utility. The histogram representation of the TMI-2 power history and initial fuel conditions used in calculations are given in Table V, along with the power histories and initial fuel conditions used in calculations of all other fuels examined here.

Air samples taken from the TMI-2 containment building environment at 7:00 a.m. on March 31, 1979, were analyzed for I and Xe activities at 8:00 p.m. on that date at Bettis Atomic Power Laboratory (BAPL), as described in Ref. 28. Reported values included a simple decay correction to 7:00 a.m., which has been removed for our use. These reconstructed 8:00 p.m. measured values are given in Table VI. We have used the TMI-2 power history and initial fuel content of Table V in tandem EPRI-CELL/CINDER-2 calculations, assuming a constant power distribution across the core. Calculated regional and core-average I and Xe activities are listed in Table VI. Isotopic ratios were formed for all isotopes of the same element from measured and calculated activities for comparison in Table VI.

Comparison of Table VI measured and calculated activity ratios substantiate the large change in the ^{133}I-to ^{133m}Xe decay branching fr tion from

14% (ENDF/B-IV) to 2.88% (ENDF/B-V). However, these measured nuclide activities must be viewed critically, since they may not represent the activities of the same nuclides produced in the fuel. Some of the initial xenon resulting from direct fission yields and iodine decay was vented to the atmosphere. Most of the remaining xenon in the containment air sample resulted only from iodine decay in the water-soluable iodides. Once the air sample was extracted, there was no subsequent formation of xenon, but there was decay, for example, of $^{133m}Xe \rightarrow {}^{133}Xe \rightarrow {}^{133}Cs$. Therefore, the time of extraction, the subsequent time of measurements, and the fractional venting of the initial Xe content are critical to calculations of relative amounts of, for example, ^{133m}Xe and ^{133}Xe. Our calculations reflect only the extraction and measurement time. We are surprised at the good agreement with calculations in view of the complex transport process.

C. H. B. Robinson-2 Samples

Assembly BO5 of H. B. Robinson-2 (HBR-2) cycles 1 and 2 was discharged on-or-about May 5, 1975. The fuel description and power history of this assembly is described in Ref. 29. Three samples of fuel were removed from rod P8 of this assembly and destructively analyzed at Batelle Columbus Laboratory (BCL) on September 24, 1975, as described in Ref. 21. Of the three samples analyzed, one has been described as atypical, because of its close proximity to a spacer grid during operation. The two remaining samples of rod P8, designated here as P8A and P8B, were taken from 12" and 68" above the bottom of the fuel, respectively.

Results of HBR-2 P8A and P8B measurements are given as atom density ratios and as burnup and exposure values determined with ASTM method E321-69. We have made iterative tandem EPRI-CELL/CINDER-2 calculations to converge on close agreement between measured and calculated atom ratios of $^{148}Nd:{}^{238}U$. Each calculation used the same histogram power history shape, constructed from the assembly-averaged power history data of Ref. 29, adjusted in magnitude to produce the desired calculated atom ratio for the sample. The beginning-of-life nuclide densities and final histogram history used for these samples are given in Table V.

The measured atom ratios, reported without uncertainties, are compared to the calculated ratios for these two samples in Table VII. Here the calculated sample burnup values are lower than those reported for the samples because of the higher ^{148}Nd net yield value resulting from the calculation. The calculated exposure values are higher than the reported values because of the higher Q values determined in the CELL calculations.

Comparison of the measured and calculated U and Pu atom fractions of Table VII shows good agreement for major nuclides. The minor constituents ^{234}U and ^{238}Pu are not in good agreement; calculated values are less than measured values by as much as 17%. The amount of ^{234}U present in a spent fuel sample is due almost entirely to the undepleted portion of ^{234}U initially present in the clean fuel. Small contributions are made from ^{235}U(n,2n) and from the decay of ^{247}Cm and ^{238}Pu. Initial fuel concentrations are generally specified simply by weight per cent ^{235}U, and ^{234}U initial concentrations must be estimated.

^{238}Pu is not initially present and is produced by three main paths. For HBR-2 sample P8B, for example, the ranking of these paths evaluated for the measurement cooling time are as follows:

1. 58% $^{235}U(n,\gamma)^{236}U(n,\gamma)^{237}U\text{-}\beta^{-}\text{-}^{237}Np(n,\gamma)^{238}Np\text{-}\beta^{-}\text{-}^{238}Pu$

2. 21% $^{238}U(n,2n)^{237}U\text{-}\beta^{-}\text{-}^{237}Np(n,\gamma)^{238}Np\text{-}\beta^{-}\text{-}^{238}Pu$

3. 21% $^{242}Cm\text{-}\alpha\text{-}^{238}Pu$

4. 0.03% $^{238}U(n,\gamma)^{239}U\text{-}\beta^{-}\text{-}^{239}Np\text{-}\beta^{-}\text{-}^{239}Pu(n,2n)^{238}Pu$.

The formation of ^{234}U and ^{238}Pu are both affected by (n,2n) reactions. The ^{238}U(n,2n) and ^{239}Pu(n,2n) cross sections are evaluated in the EPRI-CELL calculations for the temporal reactor flux, while the ^{235}U(n,2n) reaction is absent from the EPRI-CELL calculation and is evaluated from the TOAFEW-V[10] collapse of 154-group cross sections processed with a typical LWR flux.

An additional sample of HBR-2 assembly B05 fuel has recently been analyzed at Los Alamos. The sample, taken 112" above the bottom of the 144" rod E14 was not examined by standard techniques for determination of burnup. Inventories of 8 fission products and 14 actinides were measured in the determination of the rates that actinides and fission products are leached from spent fuel under controlled oxidation-reduction conditions. Iterative tandem EPRI-CELL/CINDER-2 calculations were made, using scaled variations of the assembly B05 histogram power history, to converge upon the measured $^{137}Cs/^{238}U$ atom ratio. Calculated atom volume densities (atoms/cc oxide) were converted to mass densities (atoms/gm oxide) by dividing by a density of 9.95 gms oxide/cm^3. Measured and calculated values are compared in Table VIII. The -2.88% difference from the measured ^{137}Cs and -2.80% difference from the measured ^{238}U indicates a density normaliztion problem of that magnitude.

Of the eight fission products examined, the differences between the measured and calculated concentrations of ^{154}Eu and ^{155}Eu are exceptionally large. At high exposures, the inventories of these nuclides have been produced almost entirely from multiple neutron captures on lighter fission products.

Of the fourteen actinides examined, the differences between the measured and calculated concentrations significantly exceeds the measurement uncertainty for four of the nuclides. Two of these are ^{234}U and ^{238}Pu, which have low calculated values and were discussed above. ^{240}Pu and ^{242}Pu also have calculated values significantly lower than measured values.

D. Quad Cities-1 Sample

Special test assemblies of UO_2 and mixed U-Pu oxide (MO_2) fuel were fabricated for loading in the Quad Cities-1 (QC-1)BRW core.[30,31] Fuel removed after one-cycle exposure in cycle 2 was cooled and analyzed at the G.E. Vallecitos facility.[32] Of the samples analyzed, we have selected a sample 21.5" above the bottom of the reactor fuel for EPRI-CELL/CINDER-2 modeling. Iterative tandem calculations were performed to converge upon the measure $^{148}Nd/^{238}U$ atom ratio. Calculations used a histogram power history, listed in Table V, constructed from a graphical total core power history and semi-monthly transverse irradiation probe (TIP) data indicating the relative power at a point close to the fuel sample. Because of the low elevation of the fuel sample, a 0% moderator void was used in the calculation. Measured and calculated quantities for this relatively low exposure fuel sample are compared in Table IX. Measured values were decay corrected to shutdown prior to reporting, a practice to be discouraged because of inconsistencies in nuclear data and treatment. No record is generally made of the values of data and total correction.

Differences between measured and calculated U and Pu atom fractions appear to be quite good, although many exceed the small uncertainties given. Of these, the largest difference corresponds to the low calculated value of ^{234}U. ^{238}Pu is not reported. Differences between measured and calculated Am atom fractions do not exceed the associated large uncertainties, and the agreement with Cm atom fractions is very good.

Comparisons between measured and calculated atom ratios to ^{238}U must each be examined relative to the measurement uncertainty; of these, the most alarming is the low calculated value of ^{242}Cm.

The description of the complex spectrum effects of void, burnable poisons and control-rod spaces in BWR calculations may not be adequately treated with the EPRI-CELL methodology, and EPRI has cautioned against the reliance on EPRI-CELL generated cross sections and fluxes without comparison with the results of a more complete treatment using a 2-dimensional code such as EPRI-CPM.

E. Calvert Cliffs 1 Sample

Special high-exposure test assemblies have been installed in the core of the Calvert Cliffs 1 (CC-1) PWR in a program involving the utility, EPRI, Combustion Engineering (CE), and the Safeguards Program at Los Alamos. Some of the fuel was removed after four cycles of exposure and, after cooling, analyzed at BCL. The preliminary results of measurement, currently available without uncertainties, are considered proprietary by EPRI, and the measured and calculated atom fractions and atom ratios are not given in Table X. However, EPRI has permitted our calculation and comparison of these quantities. Complete inventories for samples of adjacent rods from measurements funded by Los Alamos will soon be available for comparison without restriction.

This fuel was irradiated to high exposure in a core composed of assemblies of lower exposure. This consideration and the large water-filled control rod locations in the CE core have led EPRI to caution against the reliance of the EPRI-CELL methodology in calculating accurate exposure-dependent cross sections and fluxes. We have, however, relied upon this methodology in our calculations.

The histogram power history generated for CC-1 fuel calculations, listed in Table V, was generated from a simple full-core histogram power history presented graphically in Ref. 33. This full-core power history was scaled and used in iterative EPRI-CELL/CINDER-2 calculations converging upon the measured ^{148}Nd/^{238}U atom ratio.

Differences in measured and calculated U-atom fractions are not alarming. The 23% difference in the trace ^{235}U remaining corresponds to better than 2% agreement in the amount of ^{235}U depleted. The calculated value of ^{234}U, as before, is considerably lower than the measured value.

Differences in the remaining measured and calculated quantities are, in general, alarmingly large. In the absence of measurement uncertainties, however, it is not possible to make meaningful observations on the differences.

The high exposure fuel of CC-1 is unique. The nuclear power industry is pursuing the use of higher fuel enrichments for higher discharge exposures. The NRC is currently investigating the effects of these parameters on hypotetical accident analysis. The validity of inventory calculations for high exposure fuel has not been demonstrated beyond this work. The utilities and EPRI are encouraged to make the results of such measurements available for public benchmarking of inventory calculations.

V. CONCLUSIONS.

We have outlined the development of the popular ^{148}Nd burnup measurement procedure, and we have indicated areas of uncertainty in it and lack of clarity in its interpretation. We have examined six inventory samples of varying quality and completeness. The power histories used in the calculations have been listed for other users.

Five of the sample measurements and calculations included actinide inventories in spent fuel. The per cent difference of calculated values from measured values was determined for each sample and listed in Table XI, where fuel samples are ordered in increasing exposure. Examination of Table XI values shows that, as previously indicated, calculated inventories of ^{234}U and ^{238}Pu are routinely low. Trends are also seen in ^{240}Pu and ^{241}Pu differences, but of smaller magnitude.

We have compared calculated ratios of I and Xe isotopes with measurements of an early air sample taken from the containment building following the TMI-2 accident; these show excellent agreement.

This survey serves to illustrate the accuracy of inventory calculations for a limited number of nuclides using ENDF/B-V data. The limited range and incomplete nature of reported inventory measurements prohibits a systematic evaluation required for data adjustment recommendations and for definitive actinide and fission-product inventory uncertainties.

The Electric Power Research Institute is urged to take the lead in encouraging the cooperation between utilities, vendors, measurement laboratories, and the U. S. Nuclear Regulatory Commission in the collection and documentation of presently available and future qualified spent fuel inventory benchmarks.

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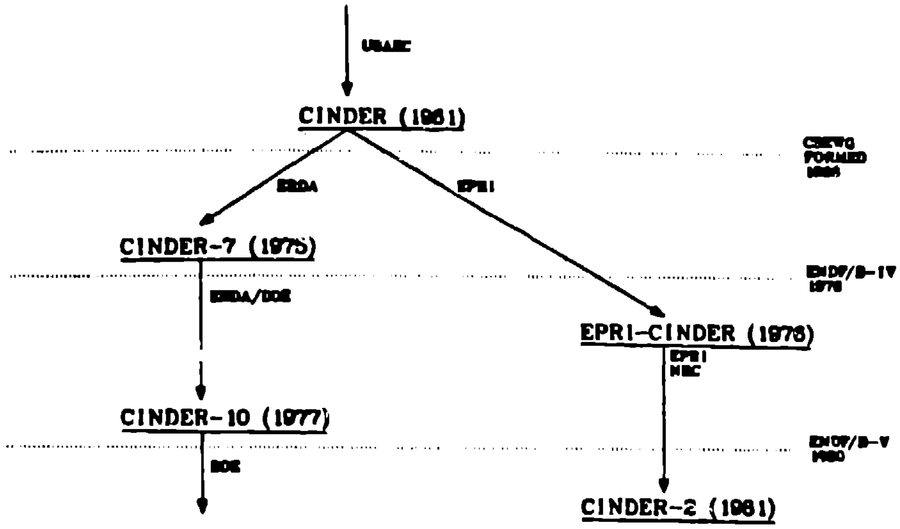

Fig. 1 Development of CINDER Codes and Libraries

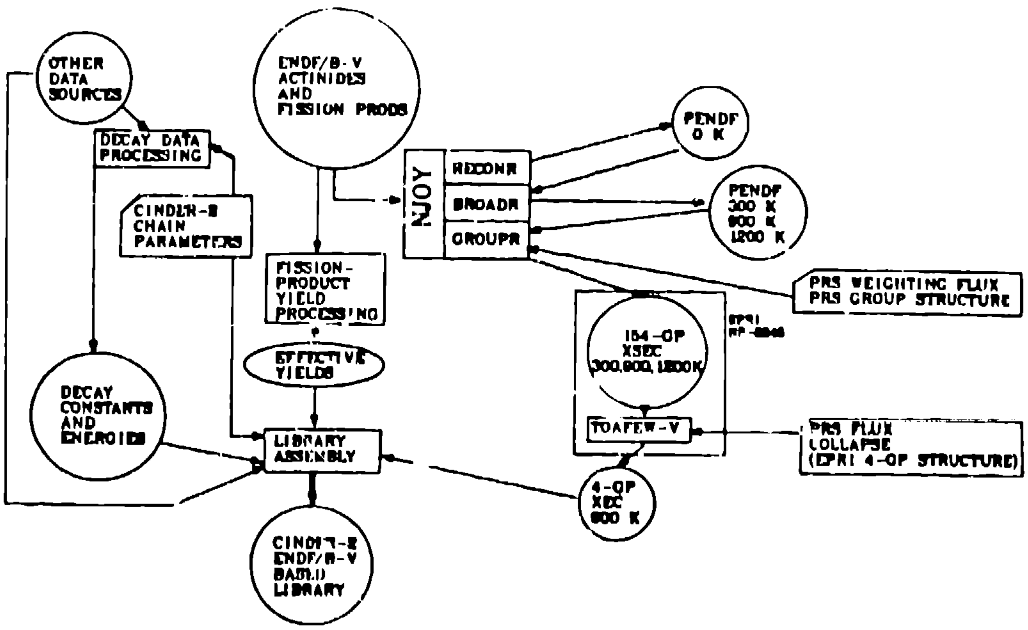

Fig. 2 Preparation of CINDER-2 ENDF/B-V Based Library

Table I

^{148}Nd Cumulative Yield Fractions

FISSIONING NUCLIDE	ENDF/B-IV	ENDF/B-V	PRELIMINARY ENDF/B-VI	ASTM E321-79
U-235(TH)	0.01690673	0.01670038	0.01674658	0.01671
U-238(FST)	0.02259347	0.02078896	0.02097547	0.02072
PU-239(TH)	0.01694488	0.01634225	0.01640564	0.01636
PU-241(TH)	0.01925721	0.01989327	0.01933803	0.02030

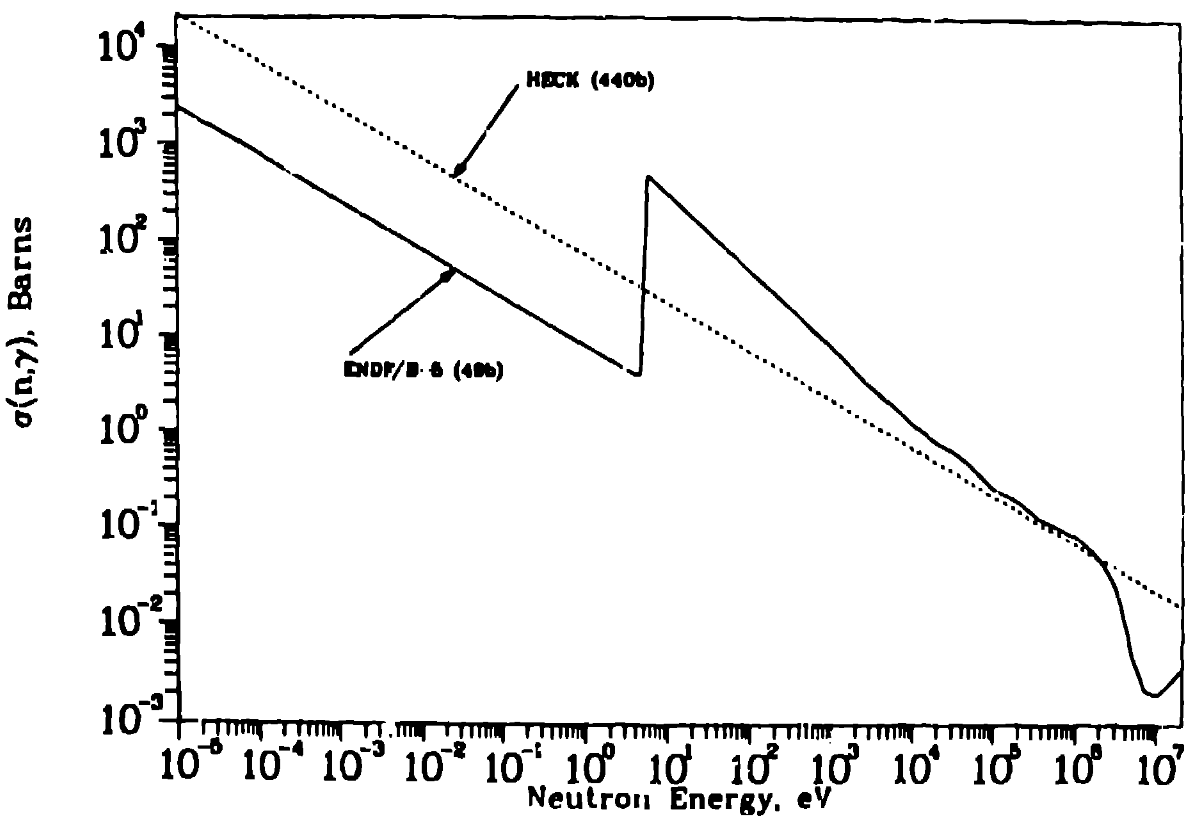

Fig. 3 Comparison of Nd-147 Cross Sections of Heck and ENDF/B-V

Table II

Calculated Production of ^{148}Nd in 2.45% U02 Calvert Cliffs 1 PWR Fuel

CALC. MWD/T	CALC. A%F	MWD/T A%F RATIO	CUMULATIVE FISSIONS #/CC	PERCENT CONTRIBUTION U235	U238	PU239	PU241	ND148 FROM MASS148 YLD #/CC	%GROSS	DIRECT YLD.%	ND148 FROM ND147(N,GAMMA) #/CC	%GROSS	GROSS ND148 #/CC	GROSS YLD.%	ND148 LOST BY ND148(N,GAMMA) #/CC	%GROSS	NET ND148 #/CC	NET YLD.%
356	0.037	9619	8.510+18	91.12	8.13	0.74	0.00	1.449+17	99.808	1.7029	2.784+14	0.192	1.452+17	1.7061	1.369+13	0.009	1.452+17	1.7060
876	0.086	9620	1.975+19	90.69	7.25	2.07	0.00	3.355+17	99.723	1.6988	9.329+14	0.277	3.365+17	1.7035	5.185+13	0.024	3.364+17	1.7031
1189	0.124	9631	2.839+19	89.91	7.09	3.00	0.00	4.819+17	99.687	1.6978	1.512+15	0.313	4.835+17	1.7031	1.724+14	0.036	4.833+17	1.7025
2941	0.305	9650	7.004+19	86.03	6.87	7.05	0.05	1.188+18	99.438	1.6957	6.712+15	0.562	1.194+18	1.7052	1.076+15	0.090	1.193+18	1.7037
4668	0.485	9675	1.113+20	82.43	6.89	10.52	0.18	1.887+18	99.329	1.6949	1.274+16	0.671	1.900+18	1.7063	2.766+15	0.146	1.897+18	1.7038
6432	0.663	9697	1.524+20	79.17	6.94	13.54	0.34	2.583+18	99.277	1.6946	1.880+16	0.723	2.602+18	1.7070	5.255+15	0.202	2.596+18	1.7035
8174	0.841	9719	1.933+20	76.22	7.01	16.18	0.59	3.276+18	99.245	1.6948	2.490+16	0.755	3.300+18	1.7076	8.564+15	0.259	3.292+18	1.7032
9915	1.018	9739	2.339+20	73.51	7.07	18.52	0.90	3.966+18	99.223	1.6952	3.105+16	0.777	3.997+18	1.7085	1.271+16	0.318	3.984+18	1.7030
11651	1.194	9756	2.744+20	70.99	7.14	20.61	1.26	4.653+18	99.206	1.6959	3.724+16	0.794	4.690+18	1.7094	1.771+16	0.378	4.673+18	1.7030
13398	1.371	9773	3.150+20	68.63	7.18	22.55	1.64	5.344+18	99.218	1.6966	4.211+16	0.782	5.386+18	1.7100	2.351+16	0.437	5.363+18	1.7025
15126	1.545	9791	3.550+20	66.47	7.25	24.23	2.05	6.027+18	99.203	1.6976	4.839+16	0.797	6.075+18	1.7112	3.031+16	0.499	6.045+18	1.7027
17208	1.754	9811	4.030+20	63.99	7.33	26.11	2.57	6.847+18	99.188	1.6990	5.602+16	0.812	6.903+18	1.7129	3.966+16	0.575	6.863+18	1.7030
19395	2.032	9837	4.669+20	60.93	7.43	28.35	3.29	7.940+18	99.198	1.7006	6.420+16	0.802	8.004+18	1.7144	5.313+16	0.664	7.951+18	1.7030
21903	2.231	9855	5.126+20	58.87	7.50	29.80	3.83	8.725+18	99.184	1.7022	7.175+16	0.816	8.797+18	1.7162	6.524+16	0.742	8.732+18	1.7034
23087	2.340	9865	5.378+20	57.77	7.54	30.56	4.13	9.159+18	99.176	1.7031	7.605+16	0.824	9.235+18	1.7172	7.248+16	0.785	9.162+18	1.7037
25614	2.591	9886	5.953+20	55.38	7.63	32.18	4.83	1.015+19	99.157	1.7051	8.627+16	0.843	1.024+19	1.7196	8.065+16	0.886	1.015+19	1.7044
26936	2.721	9897	6.254+20	54.15	7.67	32.98	5.20	1.067+19	99.164	1.7062	8.994+16	0.836	1.076+19	1.7205	1.009+17	0.938	1.066+19	1.7044
27817	2.809	9904	6.454+20	53.37	7.69	33.50	5.44	1.102+19	99.165	1.7069	9.280+16	0.835	1.111+19	1.7212	1.080+17	0.972	1.100+19	1.7045
29581	2.982	9919	6.853+20	51.87	7.75	34.46	5.92	1.171+19	99.153	1.7083	9.999+16	0.847	1.181+19	1.7229	1.235+17	1.046	1.168+19	1.7049
31356	3.157	9933	7.254+20	50.41	7.81	35.38	6.40	1.240+19	99.141	1.7098	1.074+17	0.859	1.251+19	1.7246	1.402+17	1.121	1.237+19	1.7053
33249	3.342	9948	7.680+20	48.91	7.87	36.30	6.92	1.314+19	99.130	1.7114	1.154+17	0.870	1.326+19	1.7264	1.592+17	1.201	1.310+19	1.7057
34702	3.484	9959	8.007+20	47.80	7.92	36.98	7.30	1.371+19	99.133	1.7125	1.199+17	0.867	1.383+19	1.7274	1.736+17	1.255	1.366+19	1.7058
36053	3.616	9970	8.310+20	46.80	7.96	37.58	7.66	1.424+19	99.125	1.7136	1.256+17	0.875	1.437+19	1.7287	1.888+17	1.314	1.418+19	1.7060
37568	3.764	9980	8.650+20	45.70	8.00	38.25	8.05	1.483+19	99.131	1.7148	1.301+17	0.869	1.496+19	1.7298	2.063+17	1.378	1.476+19	1.7059
39061	3.909	9991	8.984+20	44.87	8.05	38.86	8.42	1.542+19	99.124	1.7160	1.363+17	0.876	1.555+19	1.7311	2.248+17	1.445	1.533+19	1.7061
40019	4.002	9998	9.198+20	44.01	8.07	39.25	8.66	1.579+19	99.126	1.7167	1.392+17	0.874	1.593+19	1.7318	2.371+17	1.488	1.569+19	1.7061
40981	4.096	10005	9.412+20	43.37	8.10	39.62	8.90	1.616+19	99.132	1.7175	1.416+17	0.868	1.631+19	1.7325	2.496+17	1.531	1.603+19	1.7060
41058	4.103	10006	9.429+20	43.32	8.10	39.65	8.52	1.619+19	99.132	1.7174	1.418+17	0.868	1.634+19	1.7324	2.497+17	1.528	1.609+19	1.7060
42257	4.220	10014	9.698+20	42.54	8.14	40.11	9.21	1.668+19	99.133	1.7183	1.457+17	0.867	1.681+19	1.7333	2.659+17	1.582	1.654+19	1.7059
44258	4.413	10028	1.014+21	41.29	8.20	40.83	9.69	1.744+19	99.123	1.7199	1.543+17	0.877	1.760+19	1.7351	2.945+17	1.674	1.730+19	1.7060
46258	4.606	10042	1.059+21	40.08	8.25	41.51	10.15	1.822+19	99.111	1.7214	1.634+17	0.889	1.838+19	1.7368	3.247+17	1.766	1.806+19	1.7061
46835	4.662	10046	1.071+21	39.74	8.27	41.79	10.29	1.845+19	99.109	1.7218	1.658+17	0.891	1.861+19	1.7373	3.336+17	1.793	1.828+19	1.7061

Table III

K Factors from ASTM Method E 321-79

TOTAL NEUTRON FLUX (N/CM**2/S)	NEUTRON FLUENCE (N/CM**2) 1E+20	3E+20	1E+21	2E+21	3E+21
3E+12	0.9985	0.9985	0.9985	0.9985	0.9985
1E+13	0.9956	0.9952	0.9950	0.9950	0.9950
3E+13	0.9906	0.9870	0.9856	0.9853	0.9852
1E+14	0.9858	0.9716	0.9598	0.9569	0.9559
3E+14	0.9835	0.9592	0.9187	0.9008	0.8941
1E+15	0.9826	0.9526	0.8816	0.8284	0.8006

Table IV

Preliminary List of Potential LWR
Spent Fuel Nuclide Inventory Benchmarks

REACTOR	TYPE	COUNTRY	CLAD	ENRICHMENT	#SAMPLES	EXPOSURE (GWD/T) MIN	MAX
DODEWAARD	BWR	NETHERLANDS	ZR	2.5% UO2	6	0.8	2.1
GARIGLIANO	BWR	ITALY	ZR	1.6% UO2	5	9.6	14.2
				2.1% UO2	13	8.7	12.4
JPDR-1	BWR	JAPAN	ZR	2.63% UO2	30	2.2	7.0
QUAD CITIES 1	BWR	USA	ZR	2.56% UO2		11.4	
				MO2			
VAK	BWR	W. GERMANY	ZR	2.33% UO2	10	7.7	14.9
CALVERT CLIFFS 1	PWR	USA	ZR	2.45% UO2	>21	16.1	52.2
H.B. ROBINSON 2	PWR	USA	ZR	2.56% UO2	4	24.6	30.9
SAN ONOFREE 1	PWR	USA	SS	3.82% MO2	6	6.4	21.1
SAXTON	PWR	USA		0.72% MO2	69	0.1	50.9
TRINO	PWR	ITALY	SS	2.71% UO2	13	7.8	16.1
				3.13% UO2	8	7.5	18.4
				3.90% UO2	2	12.3	12.3
YANKEE ROWE	PWR	USA	ZR	2.90% UO2	33		

Table V

Power Histories Used for Spent Fuel Calculations

QUANTITY	TMI-2 AIR SAMPLE CY1, CORE TYP.		H. B. ROBINSON-2 CY1,2; ASSY.B05 ROD P8, 12"ABF		H. B. ROBINSON-2 CY1,2; ASSY.B05 ROD P8, 68"ABF		H. B. ROBINSON-2 CY1,2; ASSY B05 ROD ?14, 112"ABF		QUAD CITIES-1 CY2:ASSY.GEB-161 ROD BCG856,22"ABF		CALVERT CLIFFS-1 CY1-4;ASSY.BT03 ROD AHS024,98"ABF	
INITIAL U234/CC	<4.39033+18>		4.44528+18		4.44528+18		4.44528+18		4.40031+18		4.20591+18	
INITIAL U235/CC	<5.82902+20>		5.66790+20		5.66790+20		5.66790+20		6.90354+20		5.70028+20	
INITIAL U236/CC	<3.67315+18>		3.52605+18		3.52605+18		3.52605+18		3.67408+18		3.70686+18	
INITIAL U238/CC	<2.17845+22>		2.12845+22		2.12845+22		2.12845+22		2.21785+22		2.24019+22	
POWER HISTORY:	TIME	AVG.	TIME	AVG.	TIME	AVG.	TIME	AVG.	TIME	AVG.	TIME	AVG.
TIME STEP	HRS.	W/CC	HRS.	W/CC	HRS.	W/CC	HRS.	W/CC	HRS.	W/CC	HRS.	W/CC
1	62.00	66.68	88.5[illegible]	233.37	70.35	295.11	70.35	278.33	40.00	296.61	40.00	94.92
2	3531.50	0.00	354.11	237.80	281.39	300.53	281.39	283.47	152.00	302.40	200.00	127.97
3	315.50	57.19	663.95	237.86	527.60	300.49	527.60	283.44	360.00	226.25	475.00	101.46
4	110.00	120.51	663.95	237.80	527.60	300.43	527.60	283.41	384.00	306.00	92.00	0.0
5	178.00	0.00	892.53	237.79	703.46	300.44	703.46	283.42	288.00	233.62	807.00	127.04
6	365.00	162.33	892.53	237.69	703.46	300.34	703.46	283.33	480.00	272.53	692.00	114.18
7	105.[illegible]	0.00	744.00	280.89	741.78	300.22	741.78	283.23	360.00	128.13	1500.00	254.38
8	58.00	281.42	744.00	259.98	744.00	354.67	744.00	334.64	336.00	255.98	1500.00	253.59
9	26.50	0.00	696.00	283.91	744.00	328.31	744.00	309.75	408.00	210.49	1500.00	253.25
10	51.00	221.77	744.00	285.25	696.00	356.53	696.00	338.27	240.00	263.29	1500.00	252.95
11	636.50	0.00	853.30	289.73	744.00	360.15	744.00	339.83	360.00	206.91	1500.00	252.76
12	296.00	229.40	725.20	0.00	853.30	365.70	853.30	345.[illegible]	96.00	0.00	1497.00	252.60
13	149.00	0.00	455.99	219.33	725.20	0.00	725.20	0.00	528.00	243.61	739.00	0.00
14	233.00	269.00	893.51	218.20	455.99	278.17	455.99	262.28	1104.00	0.00	1500.00	253.41
15	320.00	248.77	744.00	270.15	893.51	275.00	893.51	253.49	384.00	122.33	1500.00	251.27
16	414.67	0.00	720.00	271.23	744.00	341.31	744.00	322.04	360.00	211.31	1758.00	251.93
17	5.33	3.84	744.00	279.46	720.00	342.58	720.00	323.21	360.00	190.26	270.00	162.91
18	672.00	268.43	720.00	244.14	744.00	352.39	744.00	332.96	504.00	211.22	392.00	147.50
19	135.75	299.11	744.00	192.[illegible]	720.00	308.36	720.00	290.94	96.00	0.00	192.00	0.00
20	16.50	0.00	744.00	156.45	744.00	241.80	744.00	228.09	624.00	127.22	484.00	127.83
21	15.75	185.08	634.22	160.61	744.00	197.84	744.00	186.60	384.00	215.00	1730.00	251.34
22	494.00	294.33	542.89	160.77	634.22	203.22	634.22	191.66	336.00	306.61	1730.00	251.55
23	52.00	COOLING	1455.30	0.00	542.89	203.19	542.89	191.64	557.00	245.04	923.00	260.51
24			967.60	147.18	1455.30	0.00	1455.30	0.00	96.00	0.00	2076.00	265.10
25			744.00	249.2[illegible]	967.60	186.30	967.80	175.66	528.00	195.89	346.00	261.17
26			744.00	215.4[illegible]	744.00	314.53	744.00	296.80	72.00	0.00	1600.00	0.00
27			720.00	224.45	744.00	272.04	744.00	256.63	240.00	183.08	815.00	242.50
28			744.00	214.17	720.00	283.57	720.00	267.53	440.00	253.69	122.00	0.00
29			720.00	162.81	744.00	270.58	744.00	255.25	480.00	272.17	753.00	250.84
30			744.00	197.93	720.00	205.83	720.00	194.14	504.00	272.[illegible]	1460.00	263.18
31			744.00	214.16	744.00	250.71	744.00	236.08	504.00	254.05	1460.00	264.73
32			672.00	220.56	744.00	270.60	744.00	255.27	480.00	251.18	1588.00	259.75
33			744.00	226.81	672.00	278.73	672.00	232.90	504.00	254.10	550.00	0.00
34			833.80	224.93	744.00	286.39	744.00	270.34			204.00	108.21
35			12162.00	COOLING	833.80	284.29	853.80	268.15			1120.00	262.81
36					12162.00	COOLING	42569.00	COOLING			1120.00	262.68
37											2064.00	0.00
38											1292.00	255.41
39											1292.00	251.68
40											1369.00	152.47
41											1369.00	152.93
42											219.00	76.77
43											1139.00	229.32
44											1642.00	265.07
45											1642.00	265.67
46											548.00	229.17
47											11232.00	COOLING
48											312.00	COOLING
49											264.00	COOLING

Table VI

Comparison of Measured and Calculated TMI-2 Containment Building Air Sample Activity Ratios

QUANTITY	MEASURED VALUE	CALCULATED VALUES: 2.01% FUEL VALUE	2.01% FUEL %DIFF.	2.67% FUEL VALUE	2.67% FUEL %DIFF.	3.00% FUEL VALUE	3.00% FUEL %DIFF.	CORE AVERAGE VALUE	CORE AVERAGE %DIFF.
BURNUP, ATOM%FISSION		0.337		0.338		0.339		0.338	
EXPOSURE, MWD/T		3265		3263		3261		3263	
SAMPLE ACTIVITIES: CURIES/LITER									
I131	[illegible]								
I133	<1.4 -5								
XE133	6.29-1								
XE133M	1.35-2								
XE135	3.00-3								
FUEL INVENTORY: CURIES/CC									
I131		5.281+0		5.223+0		5.205+0		5.235+0	
I133		8.510-1		8.537-1		8.548-1		8.532-1	
XE133		1.155+1		1.159+1		1.161+1		1.158+1	
XE133M		2.284-1		2.279-1		2.278-1		2.280-1	
XE135		4.925-2		5.030-2		5.079-2		5.014-2	
ACTIVITY RATIOS:									
XE133M:XE133	0.0214	0.01977	-8	0.01966	-8	0.01962	-8	0.01968	-8
XE135:XE133	0.0048	0.00426	-11	0.00434	-9	0.00437	-8	0.00433	-9
XE135:XE133M	0.2230	0.21564	-3	0.22072	-1	0.22300	0	0.21988	-1
I133:I131	<0.3235	0.16116	-50	0.16345	-49	0.16423	-49	0.16298	-50

AIR SAMPLES TAKEN AT 7:00 AM MARCH 31, 1979; MEASUREMENTS MADE AT BAPL AT 8:00 PM OF THE SAME DAY. REPORTED ACTIVITIES WERE DECAY-CORRECTED TO THE TIME SAMPLES WERE TAKEN. VALUES QUOTED AS MEASURED ABOVE HAVE BEEN DECAY CORRECTED BACK TO THE TIME OF MEASUREMENT.

CALCULATED VALUES GIVEN FOR THE CORRESPONDING 88-HOURS COOLING.

Table VII

Comparison of Measured and Calculated H. B. Robinson-2
2.56% PWR Spent Fuel Inventory, Cycles 1-2 Assembly B05
Rod P8 Samples 12" and 68" Above Bottom of Fuel

	SAMPLE P8A, 12"ABF			SAMPLE P8B, 68"ABF		
QUANTITY	MEASURED VALUE	CALC. VALUE	%DIFF.	MEASURED VALUE	CALC. VALUE	%DIFF.
BURNUP, ATOM%FISSION	2.559	2.526	-1.30	3.221	3.173	-1.48
EXPOSURE, MWD/T	24570	24935	+1.48	30920	31494	+1.86
ATOM FRACTIONS:						
U234/U	0.00016	0.00014	-13.53	0.00014	0.00012	-12.03
U235/U	0.00816	0.00843	+3.27	0.00612	0.00604	-1.34
U236/U	0.00326	0.00320	-1.74	0.00352	0.00354	+0.58
U238/U	0.98842	0.98823	-0.02	0.99022	0.99030	+0.01
PU238/PU	0.01143	0.00952	-16.75	0.01676	0.01407	-16.07
PU239/PU	0.59557	0.59686	+0.22	0.54261	0.54319	+0.11
PU240/PU	0.23290	0.22679	-2.63	0.25101	0.23943	-4.61
PU241/PU	0.11842	0.12291	+3.79	0.12998	0.13697	+5.38
PU242/PU	0.04168	0.04393	+5.39	0.05964	0.06635	+11.24
ATOM RATIOS:						
PU239/U238	0.00494	0.00485	-1.79	0.00518	0.00496	-4.33
ND148/U238	0.000450	0.000450	-0.01	0.000570	0.000570	+0.03

MEASURED VALUES REPORTED IN BATTELLE COLUMBUS LABORATORIES REPORT BMI-1938,P16,(1975). CALCULATED VALUES FROM THE USE OF A DETAILED POWER HISTORY, A 506.75 DAY COOLING PERIOD, AND ENDF/B-V DATA IN ITTERATIVE TANDEM EPRI-CELL/CINDER-2 CALCULATIONS TO CONVERGE UPON THE MEASURED ND148/U238 ATOM RATIO.

Table VIII

Comparison of Measured and Calculated H. B. Robinson-2 2.56% PWR Spent Fuel Inventory, Cycle 1-2, Assembly B05 Rod E14 Sample 112" Above Bottom of Fuel

QUANTITY	MEASURED VALUE	CALCULATED VALUE	%DIFF.
BURNUP, ATOM%FISSION		2.998	
EXPOSURE, MWD/T		29711	
ATOM RATIO:			
CS137/U238	0.00174	0.00174	-0.08
NUCLIDE DENSITIES, ATOMS/GM OXIDE AT 4.86 YEARS COOLING			
SR 90	2.73+18	2.37+18	-13.17
RU106	>1.71+16	2.54+16	
SB125	7.45+15	8.39+15	+12.59
CS134	7.61+16	6.92+16	-9.01
CS137	3.75+18	3.64+18	-2.88
CE144	1.41+16	1.38+16	-1.89
EU154	3.92+16	6.59+16	+67.99
EU155	1.28+16	1.83+16	+43.16
U234	3.24+17	2.71+17	-16.24
U235	1.34+19	1.40+19	+4.38
U236	7.68+18	7.31+18	-1.82
U238	2.15+21	2.09+21	-2.80
NP237	8.19+17	7.64+17	-6.69
PU238	3.25+17	2.34+17	-28.00
PU239	1.08+19	1.03+19	1.41
PU240	5.23+18	4.39+18	-16.01
PU241	2.18+18	2.11+18	-3.23
PU242	1.29+18	1.11+18	-13.57
AM241	6.55+17	6.23+17	-4.84
AM243	2.2 +17±20%	2.07+17	-6.11
CM242	1.8 +13	1.76+13	-2.23
CM244	5.1 +16±20%	4.21+16	-17.54

MEASUREMENTS BY LOS ALAMOS GROUP CNC-11; EXPERIMENTAL UNCERTAINTY ±5% UNLESS OTHERWISE INDICATED.

CALCULATED VALUES FROM THE USE OF A DETAILED POWER HISTORY, A 4.86 YEAR COOLING PERIOD, AND ENDF/B-V DATA IN ITTERATIVE TANDEM EPRI-CELL/CINDER-2 CALCULATIONS TO CONVERGE UPON THE MEASURED CS137/U238 ATOM RATIO. CALCULATED ATOMS-PER-GRAM-OXIDE QUANTITIES FROM CALCULATED ATOMS-PER-CC-OXIDE VALUES /9.95GM/CC.

Table IX

Comparison of Measured and Calculated Quad Cities-1
2.56% BWR Spent Fuel Inventory, Cycle 2, Assembly GEB-161
Rod BSG0856, Sample 21.5" Above Bottom of Fuel

QUANTITY	QC-1 BSG856 22"ABF	HBR-2 P8 12"ABF	HBR-2 E14 112"ABF	HBR-2 P8 68"ABF	CC-1 AHSO24 98"ABF
CALC. EXPOSURE, MWD/T	11837	24935	29711	31494	46836
% DIFFERENCES, (CALC.-MEAS.)/MEAS. *100					
U234/U	-7.8	-13.5	-13.9	-12.0	-9.7
U235/U	-0.5	+3.3	+7.4	-1.3	+23.0
U236/U	-0.1	-1.7	-2.1	+0.6	-2.1
U238/U	+0.01	-0.02	-0.03	+0.01	-0.04
PU238/PU	------	-16.8	-21.3	-16.1	-8.7
PU239/PU	------	+0.2	+4.2	+0.1	+7.1
PU240/PU	------	-2.6	-8.3	-4.6	-16.5
PU241/PU	------	+3.8	+5.8	+5.4	+19.6
PU242/PU	------	+5.4	-6.0	+11.7	-3.2
PU239/U238	-3.9	-0.01	-1.9	+0.03	+22.7

NOTE THAT QC-1 FUEL MEASUREMENTS DID NOT INCLUDE PU238.

Table X

Preliminary Comparison of Calvert Cliffs-1
2.45% PWR Spent Fuel Inventory, Cycles 1-4
Assembly BT-3 Rod AHS-024 Sample 98" Above Bottom of Fuel

QUANTITY	MEASURED VALUE	CALCULATED VALUE	%DIFF.
BURNUP, ATOM%FISSION	4.776	4.662	-2.39
EXPOSURE, MWD/T	45854	46836	+2.14
ATOM FRACTIONS:			
U234/U *			-9.70
U235/U *			+23.00
U236/U *			-2.10
U238/U *			-0.04
PU238/PU **			-8.70
PU239/PU **			+7.10
PU240/PU **			-16.50
PU241/PU **			+19.60
PU242/PU **			-3.20
ATOM RATIOS:			
ND143/ND148			+17.10
ND144/ND148			-1.00
ND145/ND148			+6.80
ND146/ND148			+6.70
ND148/U238			-0.16
PU239/U238			+22.70
AM241/PU239 ***			+13.70
AM243/PU239 ***			+53.20
CM242/PU239 ***			-20.50
CM244/PU239 ***			+5.50

MEASUREMENTS PERFORMED AT BATTELLE COLUMBUS LABORATORIES ON 1/18/82(*), 1/29/82(**), AND 1/05/82(***).

CALCULATED VALUES FROM THE USE OF A DETAILED TOTAL-CORE POWER HISTORY, APPROPRIATE COOLING TIMES, AND ENDF/B-V DATA IN ITTERATIVE TANDEM EPRI-CELL/CINDER-2 CALCULATIONS TO CONVERGE UPON THE MEASURED ND148/U238 ATOM RATIO.

Table XI

Comparison of Differences Between Calculated And Measured Actinide Inventories

QUANTITY	MEASURED VALUE	CALCULATED VALUE	%DIFF.
BURNUP, ATOM%FISSION	1.193	1.215	+1.8
EXPOSURE, MWD/T	11450	11837	+3.4
ATOM FRACTIONS:			
U234/U	1.776-4± 1.0%	1.638-4	-7.8
U235/U	1.512-2± 0.6%	1.505-2	-0.5
U236/U	2.063-3± 0.5%	2.061-3	-0.1
U238/U	9.861-1± 0.5%	9.827-1	+0.01
PU239/PU	7.469-1± 0.1%	7.428-1	-0.5
PU240/PU	1.810-1± 0.3%	1.901-1	+5.0
PU241/PU	6.342-2± 0.5%	5.894-2	-7.1
PU242/PU	8.694-3± 1.3%	8.154-3	-6.2
AM241/AM	7.75 -1±68.0%	6.52 -1	-15.9
AM242/AM	6.42 -3±68.0%	6.88 -3	+7.5
AM243/AM	2.18 -1±68.0%	3.41 -1	+56.4
CM242/CM	8.08 -1± 0.9%	8.05 -1	-0.4
CM243+244/CM	1.92 -1± 6.0%	1.95 -1	+1.6
ATOM RATIOS:			
ND148/U238	2.123-4± 0.67%	2.129-4	+0.3
NP237/U238	8.33 -5±18.0%	8.89 -5	+6.7
PU239/U238	3.354-3± 0.10%	3.224-3	-3.9
AM241/U238	8.98 -6±890.%	3.785-6	-57.9
CM242/U238	8.86 -7±12.9%	5.810-7	-34.4

MEASUREMENTS BY G.E., RESULTS DECAY-CORRECTED TO SHUTDOWN

CALCULATED VALUES FROM THE USE OF A DETAILED POWER HISTORY AND ENDF/B-V DATA IN ITTERATIVE TANDEM EPRI-CELL/CINDER-2 CALCULATIONS TO CONVERGE UPON THE MEASURED ND148/U238 ATOM RATIO.